Thomas Harding
Director, Licensing

CENG
a joint venture of
Constellation Energy · eDF

R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, New York 14519-9364

585.771.5200
585.771.3943 Fax
thomas.hardingjr@cengllc.com

September 22, 2010

U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTENTION: Document Control Desk

SUBJECT: **R.E. Ginna Nuclear Power Plant**
Docket No. 50-244

Transmittal of RCS Pressure and Temperature Limits Report (PTLR)

REFERENCES:

(1) Letter from T. Harding, Ginna LLC to NRC Document Control Desk, Subject: Commitment Change Associated with the Submittal of a Revised Pressure Temperature Limits Report, dated February 18, 2010

(2) Letter from T. Harding, Ginna LLC to NRC Document Control Desk, Subject: Additional Information Associated with Revised Pressure Temperature Limits Report Commitment Change, dated April 13, 2010

In accordance with the R.E. Ginna Nuclear Power Plant Improved Technical Specification 5.6.6, which requires the submittal of revisions to the PTLR, the attached report is hereby submitted.

The commitment date for submitting the attached PTLR was revised to October 1, 2010 by Reference 1. Additional information to support the revised commitment date was verbally requested by the NRC staff and provided by Reference 2.

WPLNRC-1002339

A001
NRR

There are no new commitments being made in this submittal. If you should have any questions regarding the information in this submittal, please contact Tom Harding at (585) 771-5219 or Thomas.HardingJr@cengllc.com.

Very truly yours,

Thomas L Harding

Attachment: Ginna PTLR, Revision 6

c: M. Dapas, NRC
D.V. Pickett, NRC
Resident Inspector, NRC (Ginna)

<u>Attachment</u>

<u>Ginna PTLR, Revision 6</u>

R.E. Ginna Nuclear Power Plant, LLC
September 22, 2010

CENG
a joint venture of
Constellation Energy
eDF
R.E. GINNA
NUCLEAR POWER PLANT

RCS Pressure and Temperature Limits Report
PTLR

Revision 6

Responsible Manager: [signature]

Effective Date: 9/22/10

1.0 RCS Pressure and Temperature Limits Report (PTLR)

This Pressure and Temperature Limits Report (PTLR) for the R.E. Ginna Nuclear Power Plant has been prepared in accordance with the requirements of Technical Specification 5.6.6. Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications addressed in this report are listed below:

3.4.3 RCS Pressure and Temperature (P/T) Limits

3.4.6 RCS Loops - MODE 4

3.4.7 RCS Loops - MODE 5, Loops Filled

3.4.10 Pressurizer Safety Valves

3.4.12 Low Temperature Overpressure Protection (LTOP) System

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. All changes to these limits must be developed using the NRC approved methodologies specified in Technical Specification 5.6.6. These limits have been determined such that all applicable limits of the safety analysis are met. All items that appear in capitalized type are defined in Technical Specification 1.1, Definitions. Reference 1 calculates Pressure/ Temperature Limits out to 53 EFPY.

2.1 RCS Pressure and Temperature Limits

(LCO 3.4.3)
(LCO 3.4.12)

2.1.1 The RCS temperature rate-of-change limits are:

a. A maximum heatup of 60°F per hour.

b. A maximum cooldown of 100°F per hour.

2.1.2 The RCS P/T limits for heatup and cooldown are specified by Figure PTLR - 1 and Figure PTLR - 2, respectively. These curves are based on Reference 1 as modified in Reference 12 to include instrument errors.

2.1.3 The minimum boltup temperature, using the methodology of Reference 4, Enclosure 2 is 60°F (Reference 12).

2.2 Low Temperature Overpressure Protection System Enable Temperature (Calculated in Reference 12)

(LCO 3.4.6)
(LCO 3.4.7)
(LCO 3.4.10)
(LCO 3.4.12)

2.2.1 The enable temperature for the Low Temperature Overpressure Protection System is 322°F.

2.3 Low Temperature Overpressure Protection System Setpoints

(LCO 3.4.12)

2.3.1 Pressurizer Power Operated Relief Valve Lift Setting Limits (See Reference 12)

The lift setting for the pressurizer Power Operated Relief Valves (PORVs) is ≤ 410 psig (includes instrument uncertainty).

3.0 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties. The removal schedule is provided in Table PTLR - 1. The results of these examinations shall be used to update Figure PTLR - 1 and Figure PTLR - 2.

The pressure vessel steel surveillance program (Ref. 5 as modified by Ref. 10) is in compliance with Appendix H to 10 CFR 50, entitled, "Reactor Vessel Material Surveillance Program Requirements." The material test requirements and the acceptance standard utilize the reference nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASTM E208. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Appendix G, "Fracture Toughness Criteria for Protection Against Failure," to section XI of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E185-82.

As shown by Reference 10 (Appendix D), the reactor vessel material irradiation surveillance specimens indicate that the surveillance data meets the credibility discussion presented in Regulatory Guide 1.99 Revision 2 where:

1. The capsule materials represent the limiting reactor vessel material.
2. Charpy energy vs. temperature plots scatter are small enough to permit determination of 30 ft-lb temperature and upper shelf energy unambiguously.
3. The scatter of ΔRT_{NDT} values are within the best fit scatter limits as shown on Table PTLR - 2 for the surveillance weld material. The scatter of ΔRT_{NDT} values are not within the best fit scatter limits as shown on Table PTLR - 2 for the Intermediate and Lower Shell Forging materials, which use RG 1.99 Rev. 2 Regulatory Position 1.1.
4. The Charpy specimen irradiation temperature matches the reactor vessel surface interface temperature within ± 25°F.
5. The surveillance data falls within the scatter band of the material database.

4.0 SUPPLEMENTAL DATA INFORMATION AND DATA TABLES

4.1 The RT_{PTS} value for 53 EFPY post-EPU for Ginna Station limiting beltline material is 275°F for welds and 143°F for forgings per Reference 1.

4.2 Tables

Table PTLR - 1 contains the location and schedule for the removal of surveillance capsules.

Table PTLR - 2 contains a comparison of measured surveillance material 30 ft-lb transition temperature shifts and upper shelf energy decreases with Regulatory Guide 1.99, Revision 2 predictions.

Table PTLR - 3 shows calculations of the surveillance material chemistry factors using surveillance capsule data.

Table PTLR - 4 provides the reactor vessel toughness data.

Table PTLR - 5 provides a summary of the fluence values used in the generation of the heatup and cooldown limit curves.

Table PTLR - 6 shows example calculations of the ART values at 53 EFPY for the limiting reactor vessel material.

5.0 REFERENCES

1. WCAP-17214-NP, Revision 0, "R. E. Ginna Heatup and Cooldown Limit Curves for Normal Operation and Pressurized Thermal Shock Evaluation," dated July 2010.

2. WCAP-14040-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 4, May 2004.

3. Letter from R.C. Mecredy, RG&E, to Guy S Vissing, NRC, Subject: "Application for Amendment to Facility Operating License, Revision to Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) Administrative controls Requirements," dated September 29, 1997.

4. Letter from R.C. Mecredy, RG&E, to Guy S. Vissing, NRC, "Clarifications to Proposed Low Temperature Overpressure Protection System Technical Specification," dated June 3, 1997.

5. WCAP-7254, "Rochester Gas and Electric, Robert E. Ginna Unit No. 1 Reactor Vessel Radiation Surveillance Program," May 1969.

6. Letter from R.C Mecredy, RG&E, to Guy S. Vissing, NRC, "Corrections to Proposed Low Temperature Overpressure Protection System Technical Specification," October 8, 1997.

7. WCAP-14684, "R. E. Ginna Heatup and Cooldown Limit Curves for Normal Operation," dated June 1996.

8. Letter from M. Korsnick, CEG, to US NRC Document Control Desk, Subject: R. E. Ginna Nuclear Power Plant, Licensee Amendment Request Regarding Extended Power Power Uprate. (Attachment 5 - Licensing Report), dated July 7, 2005.

9. CN-RCDA-04-149, Revision 2, "Ginna Extended Power Uprate Program Reactor Vessel Integrity Evaluations."

10. WCAP-17036-NP, Revision 1, "Analysis of Capsule N from the R. E. Ginna Reactor Vessel Radiation Surveillance Program," dated September 2010.

11. BAW-1803, Revision 1, "Correlations for Predicting the Effects of Neutron Radiation on Linde 80 Submerged-Arc Welds," dated May 1991.

12. DA-ME-08-020, Revision 2, "Pressure Temperature Limit Report (PTLR) Supporting Analysis," dated August 5, 2010.

13. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 97 to Renewed Facility Operating License No. DPR-18 R. E. Ginna Nuclear Power Plant, Docket No. 50-244.

14. LTR-AMLRS-10-26, Revision 0, "R. E. Ginna Surveillance Capsule P Withdrawal Recommendations," dated September 9, 2010.

15. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 106 to Renewed Facility Operating License No. DPR-18, "R. E. Ginna Nuclear Power Plant, LLC, Docket No. 50-244," U. S. NRC, February 23, 2009.

Material Property Basis (Reference 1)
Limiting Material: Inter to Lower Shell Forging Girth Weld and Lower Shell Forging
Limiting ART Values at 53 EFPY: 1/4T, 262°F (Circ Flaw ART), 136°F (Axial Flaw ART)
3/4T, 231°F (Circ Flaw ART), 127°F (Axial Flaw ART)

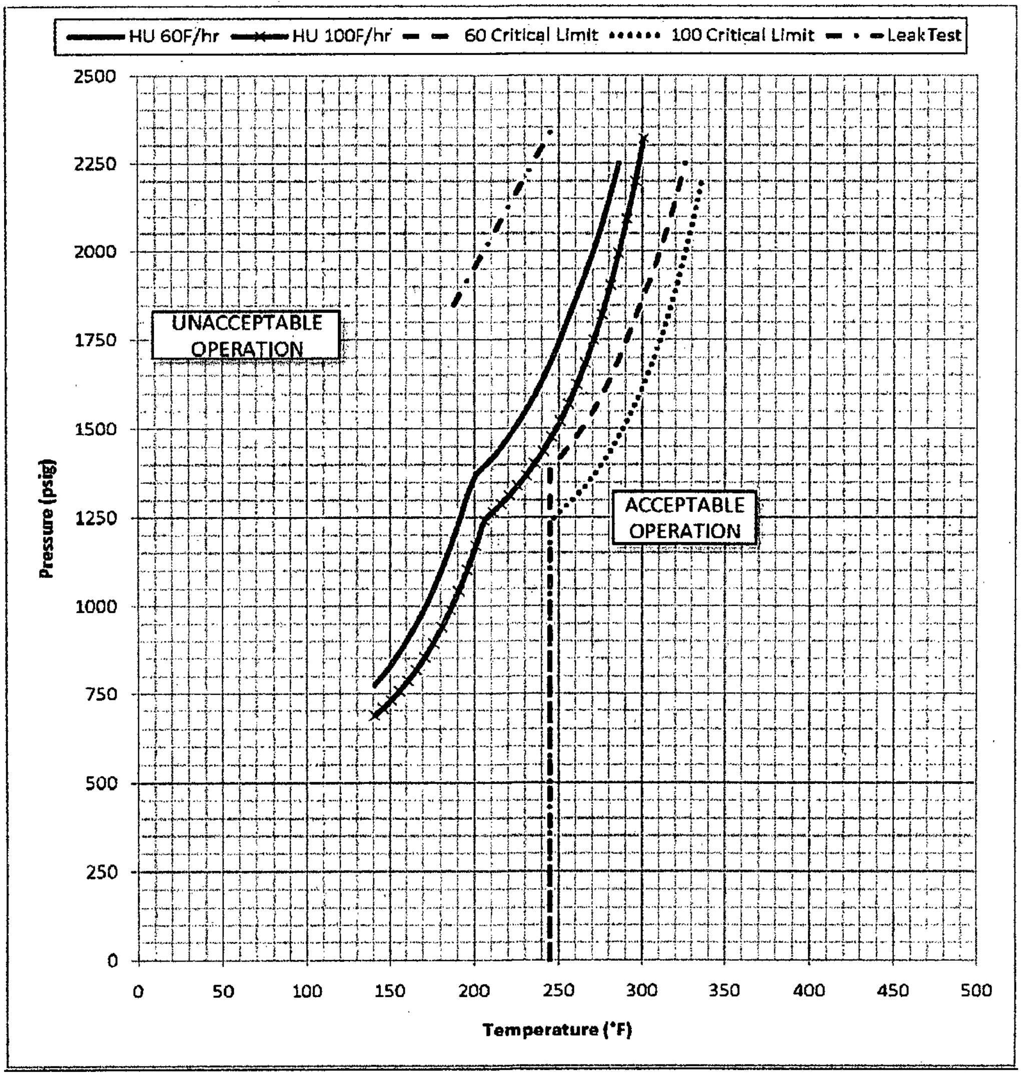

Figure PTLR - 1
R. E. Ginna Reactor Coolant System Heatup Limitations (Heatup Rates up to 100°F/hr) Applicable for the First 53 EFPY (Including Normal Instrument Errors) (Reference 12)

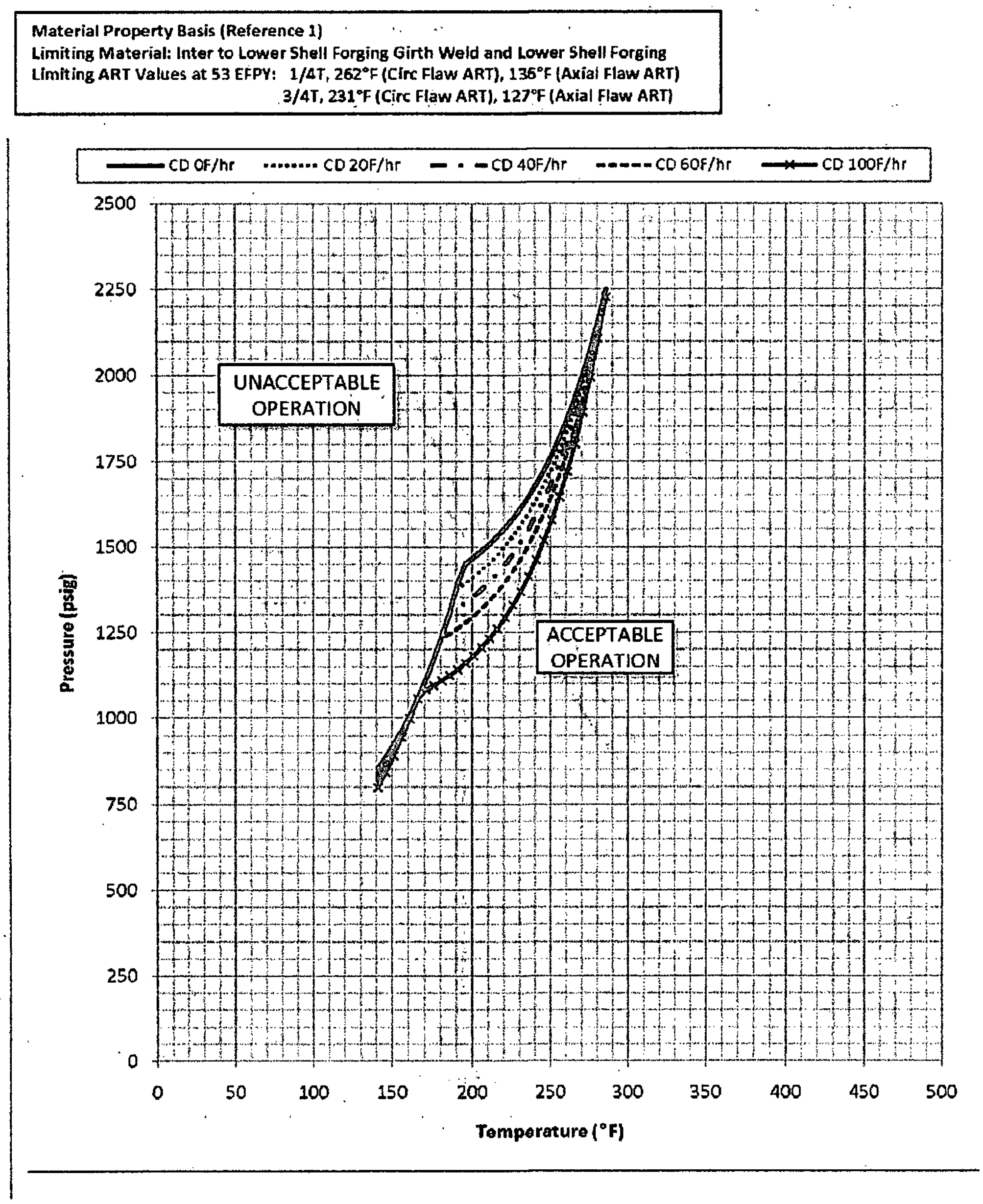

Figure PTLR - 2
R. E. Ginna Reactor Coolant System Cooldown Limitations (Cooldown Rates of up to 100°F/hr) Applicable for the First 53 EFPY (Including Normal Instrument Errors) (Reference 12)

Table PTLR - 1
Surveillance Capsule Removal Schedule[a]

Capsule	Vessel Location (deg.)	Capsule Lead Factor [b]	Removal Schedule EFPY [c]	Capsule Fluence E19(n/cm^2)[b]
V	77°	2.96	1.4 (removed)	0.587
R	257°	2.97	2.6 (removed)	1.02
T	67°	1.82	6.9 (removed)	1.69
S	57°	1.79	17 (removed)	3.64
N	237°	1.82	30.5 (removed)	5.80×10^{19}
P	247°	1.90	(d)	(d)

(a) Reference 10.

(b) Updated in Capsule N dosimetry analysis

(c) EFPY from plant startup

(d) The latest Capsule P should be removed is shortly after the vessel accumulates a fluence of 39.9 EFPY, which corresponds to a maximum 80 year fluence of 76 EFPY for the Capsule. The earliest withdrawal for Capsule P should be shortly after the vessel accumulates a fluence of 33.9 EFPY. This correlates to acceptable withdrawal for Capsule P at EOC 36, 37, 38, 39, or 40 in order to fulfill the commitment of Reference 13 to pull the final capsule shortly following accumulation of 80 years of fluence. (Reference 10 and Reference 14)

Table PTLR - 2
Surveillance Material 30 ft-lb Transition Temperature Shift

Material	Capsule	Fluence ($\times 10^{19}$ n/cm^2, E > 1.0 MeV)	30 lb-ft Transition Temperature Shift (ΔRT_{NDT})	
			Predicted[(a)] (°F)	Measured[(b)] (°F)
Lower Shell	V	0.587	26.4	34.7
	R	1.02	31.2	57.5
	T	1.69	35.5	33.6
	S	3.64	41.4	45.8
	N	5.8	44.3	91.1
Intermediate Shell	V	0.587	37.4	0.0 [(c)]
	R	1.02	44.2	20.1
	T	1.69	50.4	0.0 [(c)]
	S	3.64	58.8	76.8
	N	5.8	62.9	76.4
Weld Metal	V	0.587	135.2	146.7
	R	1.02	159.7	156.2
	T	1.69	181.8	149.7
	S	3.64	212.1	212.2
	N	5.8	227.2	216.9
HAZ Metal	V	0.587	--	30.7
	R	1.02	--	58.6
	T	1.69	--	41.0
	S	3.64	--	38.9
	N	5.8	--	107.7

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

(b) Calculated in Appendix C of Reference 10.

(c) Measured ΔRT_{NDT} value was determined to be negative, but physically a reduction should not occur, therefore a conservative value of zero is used.

Table PTLR - 3
Calculation of Chemistry Factors using R. E. Ginna and Turkey Point Surveillance Capsule Data

Material	Capsule	Capsule $f^{(a)}$	$FF^{(b)}$	$\Delta RT_{NDT}^{(c)}$	Adjusted $\Delta RT_{NDT}^{(d)}$	$FF^*\Delta RT_{NDT}$	FF^2
Intermediate Shell Forging 125S255 (L-C)	V	0.587	0.851	0.0 (e)	---	0	0.724
	R	1.02	1.006	20.10	---	20.2	1.011
	T	1.69	1.144	0.0(e)	---	0	1.31
	S	3.64	1.335	76.80	---	102.6	1.783
	N	5.8	1.430	76.40	---	109.3	2.046
					Sum:	232.1	6.875
	$CF_{125S255} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (232.1) \div (6.875) = 33.8°F$						
Lower Shell Forging 125P666 (L-C)	V	0.587	0.851	34.70	---	29.5	0.724
	R	1.02	1.006	57.50	---	57.8	1.011
	T	1.69	1.144	33.60	---	38.5	1.31
	S	3.64	1.335	45.80	---	61.2	1.783
	N	5.8	1.430	91.10	---	130.3	2.046
					Sum:	317.3	6.875
	$CF_{125P666} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (317.3) \div (6.875) = 46.2°F$						
Ginna Surveillance Weld Metal (Heat # 61782)	V	0.587	0.851	146.70	157.0	133.6	0.724
	R	1.02	1.006	156.20	167.1	168.1	1.011
	T	1.69	1.144	149.70	160.2	183.3	1.31
	S	3.64	1.335	212.20	227.1	303.2	1.783
	N	5.8	1.430	216.90	232.1	332	2.046
					Sum:	1120.2	6.875
	$CF_{Ht. \#61782} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (1120.2) \div (6.875) = 162.9°F$						

Table PTLR - 3
Calculation of Chemistry Factors using R. E. Ginna and Turkey Point Surveillance Capsule Data

Material	Capsule	Capsule $f^{(a)}$	$FF^{(b)}$	$\Delta RT_{NDT}{}^{(c)}$	Adjusted $\Delta RT_{NDT}{}^{(d)}$	$FF^*\Delta RT_{NDT}$	FF^2
Turkey Point Surveillance Weld Material (Heat # 71249)	T	0.599	0.856	163.87	147.50	126.3	0.734
	V	1.223	1.056	180.77	162.09	171.2	1.115
	X	2.897	1.282	191.06	170.98	219.2	1.644
					Sum:	516.8	3.493
	$CF_{Ht.\ \#71249} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (516.8°F) \div (3.493) = 147.9°F$						

(a) f = fluence ($x\,10^{19} n/cm^2$, E > 1.0 MeV) from Table 6-6 of Reference 10 for Ginna data and Table D-5 of Reference 10 for Turkey Point Data.

(b) FF= fluence factor = $f^{(0.28 - 0.1 * \log (f))}$

(c) ΔRT_{NDT} °F values are the measured 30 ft-lb shift values taken from Table 5-10 of Reference 10 for Ginna data and Table D-5 of Reference 10 for Turkey Point data.

(d) To address the difference in chemistry factor between the surveillance weld and the Ginna vessel weld of the same heat, the surveillance weld metal ΔRT_{NDT} values have been adjusted in accordance with Appendix D of Reference 10 using the chemistry factor ratio of: 1.07 for Heat #61782, and 0.86 for Heat #71249. The chemistry factor ratio for Heat #61782 is derived from Table 2-3 of Reference 14, and for Heat #71249 the ratio is shown in Appendix D of Reference 10. Also, adjustments were made to the measured ΔRT_{NDT} Turkey Point data to account for the operating temperature differences between the Ginna and Turkey Point vessels.

(e) Measured ΔRT_{NDT} value was determined to be negative, but physically a reduction should not occur, therefore a conservative value of zero is used.

Table PTLR - 4
Reactor Vessel Toughness Table (Unirradiated) [a]

Material Description	Cu (%)	Ni (%)	Initial RT_{NDT} (°F)
Reactor Upper Closure Head Flange	n/a	n/a	0
Intermediate Shell	.07	.69	20
Lower Shell	.05	.69	40
IS to LS Circumferential Weld	.25	.56	-4.8
Vessel Flange	n/a	n/a	-52
Nozzle Shell	0.17[b]	0.68	30
NS to IS Circumferential Weld	0.23	0.59	10

(a) Per Reference 1Table 2-1 and Table 2-2

(b) The nozzle shell forging weight-percent copper value of 0.17 was taken from Reference 15. Section 3.3, P-T Limits: Staff Evaluation, of Reference 15 states: "The staff determined that an appropriate Cu value for Ginna RPV nozzle forging should be close to 0.17 percent, the highest Cu content for the RPV shell and nozzle forgings of the entire domestic fleet based on the RVID."

Table PTLR - 5
Reactor Vessel Surface Fluence Values at 30.5 and 53 EFPY[(a)] x 10^{19}(n/cm^2, E > 1.0 MeV)

EFPY	0°	15°	30°	45°
30.5	3.20	2.01	1.45	1.31
53	5.56	3.42	2.46	2.30

(a) Reference 10 Table 6-2A

Table PTLR - 6

Calculation of Adjusted Reference Temperatures at 53 EFPY for the Limiting Reactor Vessel Material[(a)]

Parameter	Values			
Operating Time	53 EFPY			
Material	Inter. to Lower Shell Circ. Weld	Lower Shell	Inter. to Lower Shell Circ. Weld	Lower Shell
Location	1/4-T	1/4-T	3/4-T	3/4-T
Chemistry Factor (CF), °F	162.9	46.2	162.9	46.2
Fluence (f), 10^{19} n/cm^2 (E > 1.0 MeV)	3.764	3.764	1.726	1.726
Fluence Factor (FF)	1.3429	1.3429	1.1501	1.1501
ΔRT_{NDT} = CF x FF, °F	218.8	62	187.3	53.1
Initial RT_{NDT} (I), °F	-4.8	40	-4.8	40
Margin (M), °F	48.3	34	48.3	34
ART = I + (CFxFF) + M, °F	262	136	231	127

(a) Per Reference 1 Table 4-2 and 4-3